WEB-BASED DOSIMETRY AND SHIELDING CALCULATIONS IN NUCLEONICA

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ABSTRACT

The dosimetry and shielding module in Nucleonica allows the user to calculate gamma dose rates from point sources of either single nuclides or composite mixtures. The intuitive interface allows quick and accurate calculations. The present paper provides a detailed description of the module, in addition to discussing potential applications, particularly for education and training purposes.

1. Introduction

The new nuclear science web portal Nucleonica [1] has been developed at the Institute for Transuranium Elements. Nucleonica offers a suite of applications ranging from a powerful user-friendly Nuclide Explorer, which allows the user to navigate the nuclide chart and explore the properties of nuclides, to various computational and networking modules. One of the first modules developed was for dosimetry and shielding calculations for radioactive point sources and nuclide mixtures. The formalism for the dosimetry and shielding calculations is given in the Nucleonica wiki together with a detailed description [2] of the interaction of radiation with matter which provides the underlying physical basis.

2. The Dosimetry & Shielding Module

The dosimetry and shielding module allows the user to calculate gamma dose rates from point sources of either single nuclides or composite mixtures. It is possible to obtain the corresponding dose rate given a specific shield material and thickness. Alternatively, it is possible to calculate the material shielding thickness required to obtain a desired dose rate. More than 3000 nuclides and excited states with more than 53,000 gamma and x-rays are available in the Nucleonica database for dosimetry calculations, together with a choice of ten different materials for shielding purposes. The intuitive interface allows quick and accurate calculations and has been specifically designed to be suitable for use by professionals and students in nuclear science and technology.

2.1 Interface

The main interface, shown in Fig. 1 allows the users to select the nuclide, the source strength, the source/detector distance, the shield material and shield material thickness. In the example shown in Fig. 1, Co-60 ground state has been selected using the drop down menus. In the upper left hand corner, a graphic of the selected nuclide shows the half-life of the selected nuclide, 5.27 y, and information on the metastable state Co60m. The source strength can be set in different units, namely Activity (Bq), Activity (Ci), Mass (g) or number of atoms. The default value shown in Fig. 1 is 1 MBq. In addition, the user has the choice of 10 shield materials: lead, concrete (dry), tin, tungsten, uranium, water, aluminium, air (dry air at sea level) and tissue. The calculation is initiated by clicking in the Start button. The detailed results, shown in Fig. 2 in tabular form below the main interface, include the half-value layer (HVL) and the tenth value layer (TVL) thicknesses required to reduce the gamma

dose rate to 50% and 10% respectively of the initial value, and the specific gamma dose rate constant for the given nuclide.

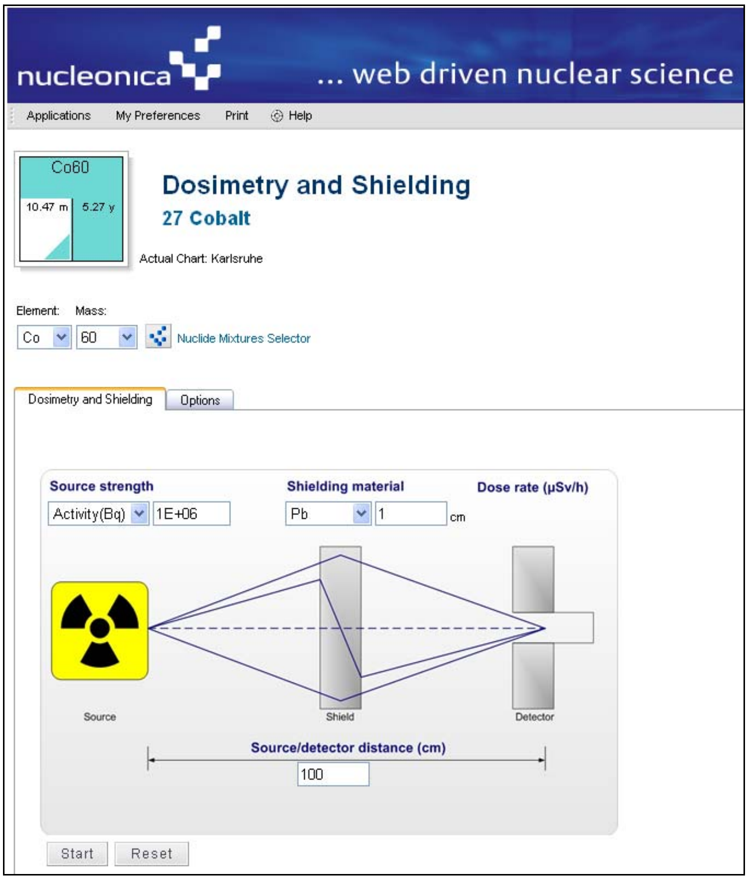

Fig 1. Dosimetry & shielding module interface.

Half-Value Shield Thickness(cm)	1.88E+00
Tenth-Value Shield Thickness(cm)	4.90E+00
Equivalent Dose Rate Constant Γ(mSv·m²/GBq/h)	3.37E-01
Gamma Dose Rate (μSv/h)	2.52E-01

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Number of lines (γ):	6	ΣE.P.(γ):	2.50E+06
Number of lines (X):	4	ΣE.P.(X):	8.35E-01
Number of lines (γ+X):	10	ΣE.P.(total):	2.50E+06

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Nuclide	Gamma Energy (MeV)	Emission Probability P (per disintegration)	Mass Attenuation Coefficient (shielding)(cm²/g)	Number of Mean Free Path(μd)	Build-up Factor	Mass Absorption Coefficient (tissue)(cm²/g)	Gamma Dose Rate(μSv/h)
27 Co 60	1.33E+00	1.00E+00	5.64E-02	6.40E-01	1.46E+00	2.89E-02	1.36E-01
27 Co 60	1.17E+00	9.99E-01	6.20E-02	7.04E-01	1.46E+00	2.98E-02	1.16E-01
27 Co 60	8.26E-01	7.60E-05	8.59E-02	9.75E-01	1.43E+00	3.16E-02	4.92E-06
27 Co 60	3.47E-01	7.50E-05	3.05E-01	3.46E+00	1.67E+00	3.21E-02	2.02E-07
27 Co 60	7.48E-03	6.44E-05	2.71E+02	3.07E+03	1	1.22E+01	0
27 Co 60	7.46E-03	3.27E-05	2.72E+02	3.09E+03	1	1.23E+01	0
27 Co 60	8.26E-03	1.31E-05	2.11E+02	2.40E+03	1	9.01E+00	0
27 Co 60	2.16E+00	1.20E-05	4.54E-02	5.15E-01	1.48E+00	2.52E-02	2.65E-06
27 Co 60	8.50E-04	1.49E-06	7.16E+03	8.12E+04	1	5.38E+03	0
27 Co 60	2.51E+00	2.00E-08	4.39E-02	4.99E-01	1.24E+00	2.40E-02	4.15E-09

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Fig 2. Detailed results for a 1 MBq Co-60 source.

This information is followed by the number of gamma and X-ray energies used in the calculation together with the quantity $\Sigma_i E_i \cdot P_i$ which is the sum of the energies multiplied by their emission probabilities. In the table in Fig. 2, the results include the contribution of each gamma-line or X-ray to the total dose rate, the mass absorption coefficients for tissue, the build-up factors, and the mass attenuation coefficients for the shield material (for refs. see [2]). The information can be re-arranged by clicking on the column headers. In the example shown in Fig. 2, clicking on the column header "Gamma Dose Rate (µSv/h)" re-arranges the table to show the main contributions to the gamma dose rate. In the case of Co-60 this is from the 1.33 and 1.17 MeV gamma lines. The information given in Fig. 2 can also be downloaded for further processing.

2.2 Basis of the Calculation

The dose rate is calculated using the point source kernel approach and is given by [3]:

$$\frac{dH(r)}{dt} = \frac{A}{4\pi r^2} \cdot \sum_i \left[E_i \cdot P_i \cdot \left(\frac{\mu}{\rho} \right)_i^{tissue} \cdot B_i \cdot \exp\left[-\left(\frac{\mu}{\rho} \right)_i^{shield} \cdot \rho d \right] \right]$$

where H(r) is the equivalent dose at distance r, A is the source activity and d is the shield thickness. The summation is over all lines i: E_i and P_i are the gamma energies and emission probabilities per disintegration, $(\mu/\rho)^{shield}$ is the mass attenuation coefficient in the shield material, $(\mu/\rho)^{tissue}$ is the mass absorption coefficient in tissue, and B_i is the dose build-up factor. In Figs 1 and 2, the calculated dose rate is 0.25 µSv/h at 1m with 1 cm Pb shielding.

2.3 Options

The Options window can be access from the appropriate tab in Fig. 1. There are two modes of operation. The user can obtain a dose rate with a given shield material and thickness. Alternatively, the thickness of shield material required to obtain a given dose rate can be calculated. The user can choose to include only gammas, X-rays, or both in the calculations. In addition, the threshold energy for contributions to the dose rate can be set by the user. The default value is 15 keV – photons with lower energy are absorbed by the outer layers of human tissue and do not contribute to the whole body dose.

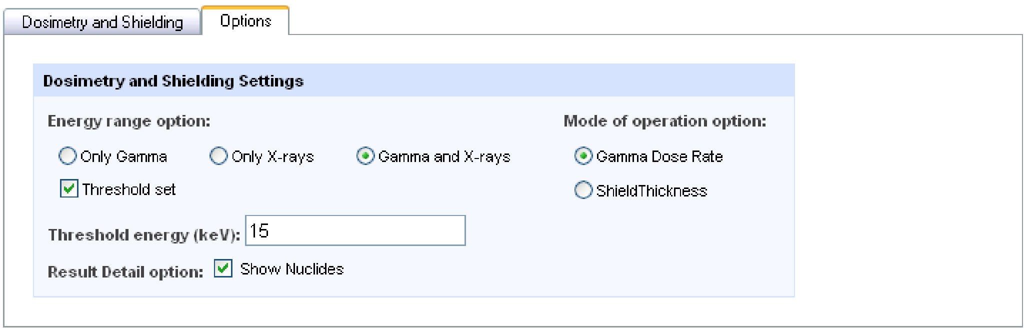


Fig 3. The Options window

3. Some Examples

3.1 Example: Occupational Exposure in Nuclear Medicine Departments

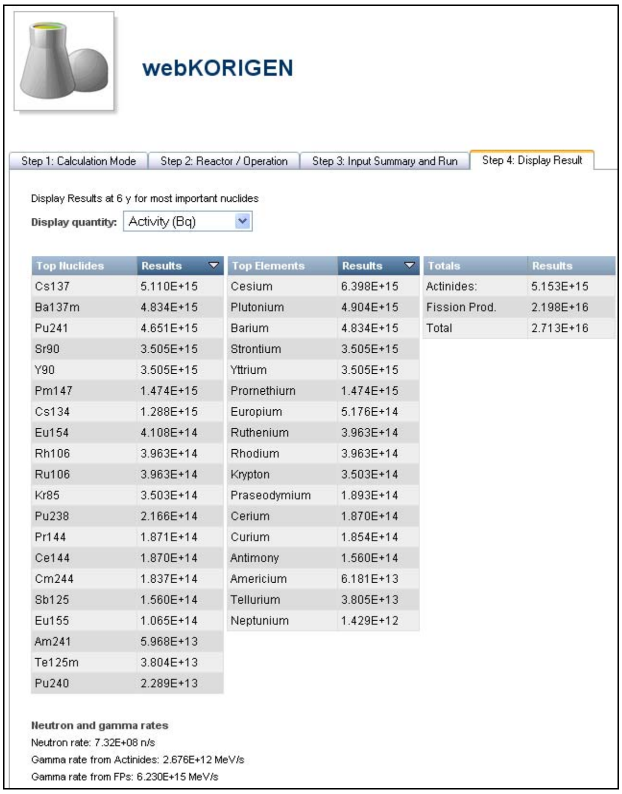
Tc-99m is a commonly used isotope in nuclear cardiology as a "tracer" for high image quality visualisation of organs. It is well suited to the role because it emits readily detectable 140 keV gamma rays, and has a short half-life of 6.01 hours. After approximately 10 half-lives, Tc-99m has almost completely decayed to its long-lived daughter Tc-99. The biological half-life of Tc-99, however, results in earlier removal from the body. A patient is injected with typically 30 mCi of Tc-99m. The treated patient must, therefore, be considered as an

unshielded source of radiation. During this time the radioactivity is present in the body, the medical staff - nurses, physicians and operators - will be exposed to radiation from the patient. It is thus interesting to estimate the dose rate received from a patient treated with 30 mCi of Tc-99m and to calculate how much shielding does a patient's body provides to protect his family and the medical staff. For this study we consider that the radioactivity is concentrated in the middle of the body (studies have shown, for example, that the technetium has tendency to concentrate in the kidneys) as a point source and shielded below 1 cm of human tissue. Calculations can then be performed with the dosimetry and shielding module of Nucleonica. The resulting gamma dose rate is 16.5 µSv/h, 65.9 µSv/h and 1.65×10^5 µSv/h at distances from the patient of 1 meter, 50 cm and 1 cm respectively. These figures are comparable with previous medical studies, see ref. [4]. If one considers that a typical procedure lasts for about 40 min, the exposure due to the patient requires the medical staff to be protected. The use of lead apron is compulsory for medical nuclear operators. From the dosimetry & shielding module, the dose rate behind a 0.5 mm lead apron at 1 m distance from the patient is reduced from 16.5 to 8 µSv/h.

3.2 Example 2: Handling of Spent Fuel in a "Hot-Cell"

The aim of this example is to demonstrate a shielding calculation with a mixture of nuclides, and to show the module working in the “shield thickness” mode. A "Hot-Cell" provides facilities for performing operations on highly radioactive material with minimal radiation exposure to the personnel involved. Most hot cells are designed to accept fuel rods for post irradiation investigations. To protect the user against gamma radiation, the different cells are shielded with lead bricks and can be operated through a lead glass window.

For this study we create a nuclide mixture, using the mixture option in Nucleonica, based on the main contributors to the gamma heat calculated using a webKORIGEN calculation of a spent fuel rod 4.2% enriched from a standard PWR reactor (50GWd/t) and after 6 years of cooling. The total mass of the 20 most important contributors to the total activity is 3.4 g per kg of spent fuel. The maximum mass load of spend fuel introduced into the hot cell is typically about 500g.


webKORIGEN

Step 1: Calculation Mode | Step 2: Reactor / Operation | Step 3: Input Summary and Run | Step 4: Display Result

Display Results at 6 y for most important nuclides

Display quantity: Activity (Bq)

Top Nuclides	Results	Top Elements	Results	Totals	Results
Cs137	5.110E+15	Cesium	6.398E+15	Actinides:	5.153E+15
Ba137m	4.834E+15	Plutonium	4.904E+15	Fission Prod.	2.198E+16
Pu241	4.651E+15	Barium	4.834E+15	Total	2.713E+16
Sr90	3.505E+15	Strontium	3.505E+15		
Y90	3.505E+15	Yttrium	3.505E+15		
Pm147	1.474E+15	Promethium	1.474E+15		
Cs134	1.288E+15	Europium	5.176E+14		
Eu154	4.108E+14	Ruthenium	3.963E+14		
Rh106	3.963E+14	Rhodium	3.963E+14		
Ru106	3.963E+14	Krypton	3.503E+14		
Kr85	3.503E+14	Praseodymium	1.893E+14		
Pu238	2.166E+14	Cerium	1.870E+14		
Pr144	1.871E+14	Curium	1.854E+14		
Ce144	1.870E+14	Antimony	1.560E+14		
Cm244	1.837E+14	Americium	6.181E+13		
Sb125	1.560E+14	Tellurium	3.805E+13		
Eu155	1.065E+14	Neptunium	1.429E+12		
Am241	5.968E+13				
Te125m	3.804E+13				
Pu240	2.289E+13				

Neutron and gamma rates
Neutron rate: 7.32E+08 n/s
Gamma rate from Actinides: 2.676E+12 MeV/s
Gamma rate from FPs: 6.230E+15 MeV/s

Fig 4. Results of a webKORIGEN calculation for the most important contributors to the gamma heat from spent fuel.

German regulations give a limit for occupationally exposed workers (category A) of 20 mSv/year assuming 2000 working hours (i.e. 2000 h of exposure). This leads to an hourly dose rate of 10 µSv/h for a regular worker working. The limit for occupationally exposed workers (category B) is 6 mSv/year assuming 2000 working hours (2000 h of exposure) hence equivalent to 3 µSv/h. We can then use the dosimetry and shielding module and using the “shield thickness” mode calculate the thickness of lead required to obtain 3 µSv/h at 1 m of distance for 1.7 g of the calculated mixture obtained with webKORIGEN. The calculation gives a lead thickness of 14.8 cm. In most of the existing hot cells, reinforced concrete walls provide the main shielding used for construction. The module has dry concrete as a shield option. This can then be used as an approximation for the concrete walls. The calculation gives a required thickness of 92.8 cm of concrete, as shown in Fig. 5. Typically, hot-cells have wall thickness of approximately 1 m of reinforced concrete and 1 m thick lead glass windows.

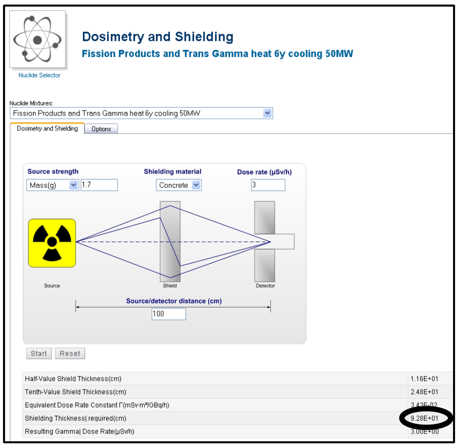


Fig 5. Calculation for the concrete shielding requirement for 1.7 g of the main transuranics elements and fission products, corresponding to 500g spent fuel.

4. Future developments

The Dosimetry & Shielding module in Nucleonica is under continuous development driven in particular by user demand. Future upgrades will include additional shielding materials, multilayer shielding options, and volumetric source dosimetry based on Monte Carlo calculations.

5. Conclusion

The dosimetry and shielding module in Nucleonica is a versatile tool for quick and accurate dosimetry and shielding calculations. It allows the user to calculate gamma dose rates from point sources of single nuclide and mixtures, through a choice of 10 different shield materials. Over 3000 nuclides with more than 53000 gamma lines are available in the database. The examples described provide an overview of the features available and the flexibility of the module for education and training purposes in nuclear science.

6. References

[1] J. Magill “Nucleonica: a Web Portal for the Nuclear Sciences”, this conference. See also the Nucleonica website at www.nucleonica.net.

[2] The Dosimetry & Shielding Manual, see the nucleonica wiki at: http://www.nucleonica.net:81/wiki/index.php/Help:Dosimetry_%26_Shielding.

[3] J. Magill, Nuclides.net, Springer-Verlag, Heidelberg, 2003.

[4] M.C. Limacher et al., J. Am. Coll. Cardiol., 31, 1998, 892-913. http://content.onlinejacc.org/cgi/reprint/31/4/892